May 1, 1990

Docket No. STN 50-605

Patrick W. Marriott, Manager
Licensing & Consulting Services
GE Nuclear Energy
General Electric Company
175 Curtner Avenue
San Jose, California 95125

Dear Mr. Marriott:

SUBJECT: REQUEST FOR ADDITIONAL INFORMATION REGARDING THE GENERAL ELECTRIC COMPANY APPLICATION FOR CERTIFICATION OF THE ABWR DESIGN

During the course of the review of your application for certification of your Advanced Boiling Water Reactor Design, we have identified a need for additional information. Our request for additional information, contained in the enclosure, addresses Chapter 19 of the SSAR relating to severe accidents. We request that you provide your responses to this request by May 30, 1990. If you have any concerns regarding this request please call me on (301) 492-1104.

Sincerely,
/s/
Dino C. Scaletti, Project Manager
Standardization and Life
Extension Project Directorate
Division of Reactor Projects - III, IV, V
and Special Projects
Office of Nuclear Reactor Regulation

Enclosure:
As stated
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UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D. C. 20555

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ENCOLSURE

ABWR SEVERE ACCIDENT REVIEW

725.62. According to Response 5 of GE's response to previous staff questions, all the Residual Heat Removal (RHR) pumps will start automatically upon receipt of low water level signal or high drywell pressure signal and can be transferred to other operating modes while they are running. Is the transfer of the RHR pump flow from injection mode (referred to "V") to the containment heat removal mode (referred to "W") done automatically without requiring any operator actions? If so, provide discussions regarding modelling aspect of operator actions for the containment heat removal mode of the RHR system.

725.63. For scenario involving vessel isolation event followed by the failure of the High Pressure Core Flooders (HPCF), Reactor Core Isolation Cooling (RCIC) System and RWCU System, and successful vessel depressurization, will both "V" function and "W" functions be required simultaneously for successful core cooling (during the mission time considered) and long-term heat removal? If so, state the minimum trains of the RHR system needed to avoid a core damage.

725.64. By definition of "Class 2 Sequences," the containment heat removal systems (RHR system) have failed following a transient and a postulated LOCA event. Therefore, provide discussions regarding adequacy of crediting the RHR system (such as fast recovery) for the scenario involving a vessel isolation event followed by the failure of the HPCF system, the RCIC system, and successful vessel depressurization with coolant injection only achievable by the LPFL mode of RHR. If the RHR system can be used (during this scenario) for both "V" and "W" functions, can train A of the RHR system alone perform both "V" and "W" functions to avoid a core damage?

725.65. The staff notes that the pumping capacity of the RHR pumps of the ABWR design is lower than that of the operating BWR designs. Therefore, provide discussions regarding the modelling adequacy of the RHR system (use of one of three RHR trains to maintain the pool temperature below the heat capacity-temperature limit) for the scenario involving the vessel isolation event followed by a fail-to-scram event. GE's discussions should include supporting pool temperature calculations, including the assumed amount of heat dump to the pool following the above scenario.

725.66. The staff believes that a gas turbine-generator (in addition to the three train diesel generator system) added to the ABWR design will reduce the frequency of sequences involving early core damage following a loss of offsite power event with a postulated common mode failure of the diesels. Thus, provide discussions for the following:

a. What is GE's definition for the black-start capability for the gas turbine-generator?

b. Will the gas turbine-generator be started automatically?

c. If a start failure of the gas turbine-generator will occur, can it be started from the main control room?

d. Does the operator have to decide as to which class 1E 4.16kV bus should receive ac power generated by the gas turbine-generator?

e. Did GE perform a trade-off study involving the benefits of a seismically qualified gas turbine-generator.

f. What are assumptions made in quantifying the results provided in Table 19.3-6 of the ABWR PRA (Amendment 9) which includes the impact of adding a gas turbine-generator? In particular, were the initiating event frequencies (such as BE2, BE8, BE0, TE2, TE8, TE0) recalculated by modifying the event tree provided in Figure 19D.4-4? If so, provide these estimates. Also, provide, for the case of adding a gas turbine-generator, similar results provided in Tables 19D.4-1 and 19D.4-3.

725.67. Provide discussions related to the use of the RCIC system unavailability estimate documented in Table 19D.4-1 (under the column of offsite power event), in the event tree quantification. Also, provide statements related to the consistency of the RCIC system unavailability estimate used for the quantification of the ATWS event tree, and the corresponding estimate documented in the Table 19D.4-1.

(II) Seismic Review Questions

725.68. Provide the scientific details of seismic hazard analyses performed for the ABWR Design review and the basis for selection of the seismic hazard curve (Figure 19.4-2). The discussion should include site seismicity characterization of various (five reference sites) sites considered in eastern United States of America, including the combination method used to develop a single enveloping seismicity hazard curve to represent an enveloping site to locate the ABWR design, and the associated uncertainty estimates for the use of a single seismicity hazard curve. The discussion of the site characterization should include critical site parameters such as soil-structure interaction for various sites considered. There are some seismic terms used in GE's seismic risk analysis which are confusing to the staff. What is the parameter used for describing the seismic hazard and fragility? For example, it is variously used to represent as the effective peak ground (Figure 19.4-2) and mean peak ground acceleration (Section 19.4.3.2.1).

725.69. Provide the ABWR-specific fragility calculations for the following structures and component: Containment, Reactor Building, Main Control Room (including control room suspended ceilings, if any), Reactor Pressure Vessel (RPV), RPV Pedestal, RPV Shroud Support, CRD Guide Tubes, CRD Housings, Fuel Assemblies, Containment Vent System, Passive Flooder, SRV Pipes to Suppression Pool. If generic component fragilities have been used, provide a detailed discussion how the generic component fragilities were assigned. The discussion should include also applicability of the uncertainty estimates due to variations in ABWR design-specific component design may have. Does the failure mode, "Relay Chattering," applicable to the ABWR design? If so, provide discussions regarding the modelling of electrical equipment (such as breaker) to account for relay chattering effect in fragility quantification. Provide also discussions regarding sequences (such as loss of containment isolation function) that could result from relay chattering failure mode, and method of quantifying such failure modes (including human recovery actions involved, if any). Provide the details regarding the seismic capacity of the fire protection system (including the valves F005, A, B, C of the ac independent fire water system. Provide also the seismic capacity of small piping (if used) and valves (14 and 22 inches in size) of the containment overpressure relief (COR) system, addressing the failure mode, "Normally open valves fail closed" and including human recovery actions involved, if any.

725.70. The staff understands that the seismic PRA performed for the ABWR design is limited in nature due to the design stage (FDA). However, our past seismic risk review experience indicates that seismic risk profiles of as-built-ABWR plant in U.S. could be different due to variations in construction standards by various architects. Therefore, provide discussions regarding the construction interface requirements such as allocated fragility estimates for all applicable mechanical and electrical component of the ABWR design, as practicable, including the severe-accident design basis and/or goals on which allocation of such fragility estimates will be performed. These discussions should also include consistency between requirements outlined in Electric Power Research Institute (EPRI) - Advanced Light Water Reactor (ALWR) Requirements Document, and design requirements to be proposed to various architects by GE.

725.71. Provide ABWR-specific layout drawings (in larger size) which show clearly major structures and equipment. Provide also as-designed structural drawings which show the details of the RPV support arrangement, RPV internals arrangement, drywell and the reactor building.

725.72. Provide a copy of the ABWR PRA seismic input data such as seismic hazard curve and seismic accident sequences applicable to the ABWR design, in the form of a hard copy (tabular forms and boolean equations) as well as a magnetic media. These data are needed to facilitate staff's audit review.

725.73. The staff believes that the determination of a particular seismic intensity (for risk modelling purposes) at which evacuation scheme at a particular site following a postulated severe-accident will impact greatly the risk estimates (early fatality estimates). Provide discussions regarding the determination of the break point of the seismic intensity (in terms of EPG) at which evacuation were considered impossible for ABWR risk estimation purposes.

725.74. Our past PRA review experience indicates that fires and internal floods contribute significantly to the overall core damage frequency at nuclear power plants. The staff also believes that, with respect to the ABWR design protection against fires and internal floods, GE will provide significant design improvements to current separation requirements and divisional (redundancy) requirements related to all safety systems and components. Nevertheless, the ABWR PRA (Amendment 9) has not documented the core damage frequency analysis of fires and internal floods. Therefore, provide the results of screening analysis (including the screening criteria) performed for the ABWR design to show that fires (panel fires, transient combustible fires, cable fires) and room-specific floods do not significantly contribute to the overall core damage frequency. Provide also statements regarding consistency between requirements outlined in the ALWR Requirements Document and current ABWR design requirements related to fire protection and flood protection schemes.

725.75. In developing the fault trees for seismically induced failure of the ECCS, such as HPCF, RCIC, LPCF and RHR (Figures 19I.2-1 through Figure 19I.2-4 of the ABWR PRA), no explicit modeling of the dependence of these ECCS on electric power or service water system was made. Nevertheless, fault trees were developed in Figure 19I.2-6 and Figure 19I.2-7 to depict seismically induced failure of Division 1 service water and seismically induced failure of Division 1 electrical power respectively. Please explain how the latter two fault trees developed for the support system were combined with event tree top events to generate minimal cut sets for seismic core damage sequences.

725.76. Following loss of offsite power due to seismic events, an important subsequent concern is whether or not emergency power and service water are available. Failure of emergency power (diesels or gas turbine generator) and failure of service water system may be considered as two virtually independent events. In the seismic event tree (Figure 19I.3-1), however, these two events are combined together and treated as a single event tree top event, PW. Please explain how the failure probability of this top event was estimated. Was the gas turbine generator included in evaluating the availability of emergency power?

725.77. Were random failures of the ECCS, such as HPCF, RCIC, LPCF and RHR, taken into account in the quantifications of seismic core damage frequency? If so, please provide a list of random failure probabilities for the important systems and components used in the quantifications.

725.78. On page 19.4-11 of the ABWR PRA (second paragraph), it is stated that "Since these fault trees (meaning those shown in Appendix I) are specifically for evaluation of seismically-induced failures, only those components vulnerable to seismic failure are included in the trees." In reality, however, those fault trees also contain basic events (depicted with an "X"), that would not occur as a result of an earthquake. Please explain the contradiction.

725.79. Please provide justification of considering heat exchanger failure in the RHR and service water fault trees, while ignoring it in the fault trees of RCIC, HPCF and LPCF. 13. In the fault tree developed for service water system (Figure 19I.2-6), the motor-operated valve, WMVS3DH, is considered seismically vulnerable, while three other similar motor-operated valves are considered seismically invulnerable. What is the basis for making such a distinction? In the fault tree depicting seismically induced failures of RCIC (Figure 19I.2-2), three identical basic events are used to denote non-seismic failure of an isolation valve (MOV). Are these three basic events intended for failures of three different isolation valves?

725.80. Please provide a terse but systematic description of how the Boolean expressions derived from the seismic event trees and fault trees are combined with seismic hazard function, component and structure fragilities and other unavailability data, and integrated to obtain the frequency of individual accident sequences.

725.81. For ATWS events with failure to initiate SLC, what alternative means are available for injecting boron in order to shut down the reactor? What failure probability was used in the sequence frequency quantification for the event tree top event, FCTR (flow control/alternate boron), appearing in Figures 19I.3-1, 3-3 and 3-4?

725.82. The event tree top event, W1, appearing in Figures 19I.3-2, 3-3 and 3-4, is defined to be "at least one RHR." How many trains (1, 2 or 3) of RHR were actually used in the sequence frequency quantifications? Please also list the random failure probability assigned to this event in each figure.

725.83. In the seismic event tree, Figure 19I.3-1, credit is given to fire water (event tree top event, FA) for the following transient scenarios: (a) station blackout, successful scram, failure of RCIC; (b) station blackout, successful scram and RCIC; and (c) station blackout, failure of scram but successful RCIC. What is the unavailability of fire water system in each case?

725.84. In the seismic ATWS event tree, Figure 19I.3-3, the last sequence involves failure of SRVs to open following the inception of an LOOP ATWS. Please explain why this sequence is classified as Class IC, which, by definition, involves low pressure vessel failure. Please also clarify the description of accident classes for Class IV-1 (ATWS with one injection pump) and Classes IV-2, 3, 5 (ATWS with multiple injection pumps) in connection with the relevant sequence classifications performed in Figure 19I.3-3. What is the basis of choosing 2, 3 or 5?

725.85. The suppression pool drain accidents due to RHR pipe break are considered to be 00SN, which implies no fission product release. However, if the suppression pool is drained, the passive flooder is not operable, and therefore extensive CCI will continue. Why is this effect not considered in determining the fission product release for this sequence?

725.86. The firewater availability is considered to be 0.9 for the vessel cooling except for the Class IB-2 accidents, where it is 0.999. The firewater availability for the drywell spray is also assumed to be 0.999 (page 19J.4-1). However, the firewater availability in the internal event analysis was assumed to be 0.9 for vessel cooling and 0.99 for the drywell spray. Why these are substantially more reliable for the seismic events?

725.87. On page 19J.3-1, it is stated that "ARC" is solely due to firewater. Why then is the "ARC" Yes branch fraction not 0.999 in the CET's (why 0.944)?

725.88. How are the "CHR" branch fractions evaluated?

725.89. In Figure 19J.5-7, Sequence 4 was binned as NSRCFSDL. Shouldn't this be binned as OK, since this sequence represents continued core cooling by firewater? (Compare this with Figure 19J.5-6 for Class II.) Why are the "CHR" No and "CC" Yes branch fractions of Class IV not same with those of Classes II?

725.90. The firewater availability is assumed to be 0.9 for Class IV (Figure 19J.5-7). However, it was stated in the internal event analysis that no credit was taken for the firewater system to prevent core damage for Class IV because the stability of the reactor during an ATWS has not been examined (page 19D.5-10). Please clarify.

725.91. It appears that the loss of transformer contributes significantly to 'loss of core cooling' accidents. Why isn't this considered to be a blackout sequence (IB)? What fraction of Class IA is due to this scenario and what fraction is due to other causes such as loss of injection pumps or lines, etc? What is the RHR recovery probability for each of these sequences?

725.92. It is stated on page 19J.4-1 that the reliability used for the firewater system is also used for the transformer bypass operation. Does the "ARV" Yes branch fraction take this high reliability into consideration?

725.93. Why is RHR assumed to be lost for the Class IA accidents? (In the internal events CET for Class IA, the RHR availability was 0.99.) Do all accidents in SCET assume loss of power due to loss of transformer and require the bypass of the transformer?

725.94. Is the loss of the offsite power by seismic events with subsequent failure of onsite power considered to be IB-2? Does Class IB-2 include the loss of power due to loss of transformers? What is the RHR recovery probability for each of these sequences?

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